Phase 2 Topical Report: DE-FG07-99ID13780, "Investigation of Minimum Film Boiling Phenomena on Fuel Rods Under Blowdown Cooling Conditions "

PROJECT TITLE:	Investigation of Minimum Film Boiling Phenomena on Fuel Rods Under Blowdown Cooling Conditions
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Introduction

The main objectives of "Investigation of Minimum Film Boiling Phenomena on Fuel Rods Under Blowdown Cooling Conditions" are to obtain minimum film boiling temperatures and heat transfer coefficients typical of the blowdown cooling period of a large break loss of coolant accident (LOCA) in the hot assembly of a light water reactor. Flow through the hot assembly during blowdown cooling is expected to be a post-CHF dispersed droplet flow at moderate pressure, 75 to approximately 400 psia (0.517 to 2.76 MPa). Segments of the hot assembly may quench, if surface temperatures drop below the minimum film boiling temperature. The minimum film boiling temperature is dependent on several parameters including flow rate, pressure, flow quality and the clad surface. Thus, the experimental goal of this investigation is to measure rod temperatures and determine surface temperatures and heat transfer coefficients for these prototypical blowdown conditions for rods with clad materials similar to actual fuel rods. The goal of the analytical part of this investigation is to develop a mechanistic model for the quench process in a blowdown cooling flow.

The objectives of the second phase of this investigation were to perform initial data collection and establish the experimental techniques and procedures for obtaining additional, high quality data. In particular, the second phase was intended to : (a) complete facility construction and obtain experimental data for Inconel and Zr clad heater rods, and (b) conduct tests using neutron radiography to visualize droplet flows within titanium tubes.

Experimental Facility and Procedure

At the end of Phase 1, the experimental facility had been designed, and major components had been acquired. Construction was completed in early June, and "shake-down" testing was then started to examine the power and steam supply systems, check and calibrate the instrumentation and program the data-acquisition system   Figures 1 and 2 show schematics of the test facility, and Figure 3 shows a photograph of the test vessel.

These shake-down tasks were accomplished in June 2000, and initial testing began in July 2000 with an Inconel clad rod installed in the test facility. Tests were conducted by first heating the nuclear rod simulator to a temperature above the saturation temperature corresponding to the desired test pressure, with no steam flow to the test section. This heated the rod, the inner Inconel tube, and the relatively large outer housing and eventually, steady-state conditions were attained. Steam from the high pressure steam supply was then allowed to flow into the test vessel and pressurize the vessel. (Pressurization of the vessel usually takes 5 to 15 minutes.) The rig was allowed to reach a new steady-state, which was verified by heater rod, tube wall, inner housing, and steam probe temperatures. If the heater rod and tube wall temperatures were less than that expected for Tmin, the heater rod power was increased and a new steady-state allowed to be reached.

While the steam flow and rod temperatures were rising to their steady state values, distilled water was pumped through the coolant heat exchanger and heated to a specified subcooling. Until testing started, water from the heat exchanger bypassed the test vessel and was dumped into the rig exhaust line. When operating conditions had been established in the rig and coolant loop, the control valve was opened and subcooled water was allowed to flow into the upper plenum of test section where it could become entrained in the steam flow. (Note: The upper plenum design was such that the injected steam and water was forced through the concentric annulus between the heater rod and the inner tube in Figure 2. There is no flow between the inner tube and the test vessel housing. Small bleed holes in the lower support plate allow steam to fill the tube - housing region and equalize pressure.) Transient measurements, which included heater rod thermocouples, thermocouples attached to the Inconel tube and housing wall, steam probe temperatures, steam flow rate and plenum pressure were recorded by an HP 34790 data acquisition system.

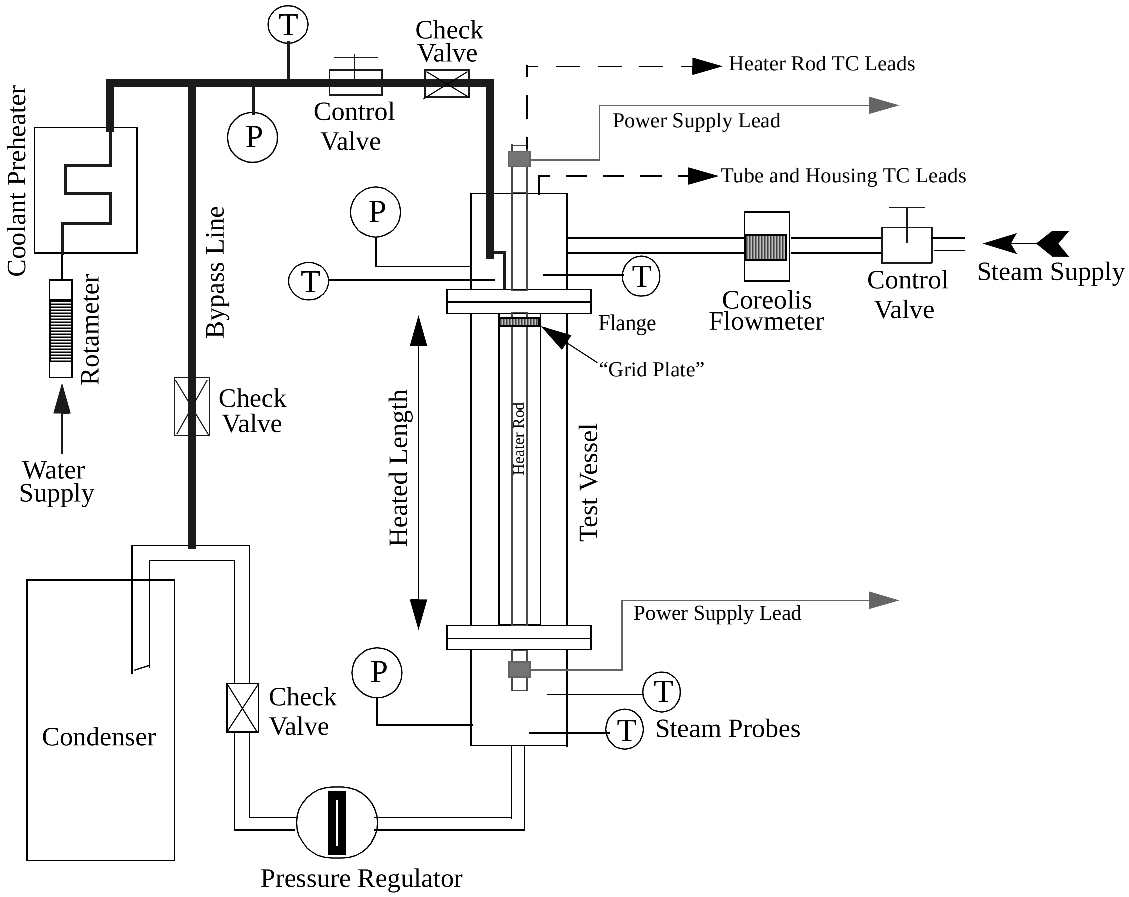

Figure 1. Test Facility Schematic

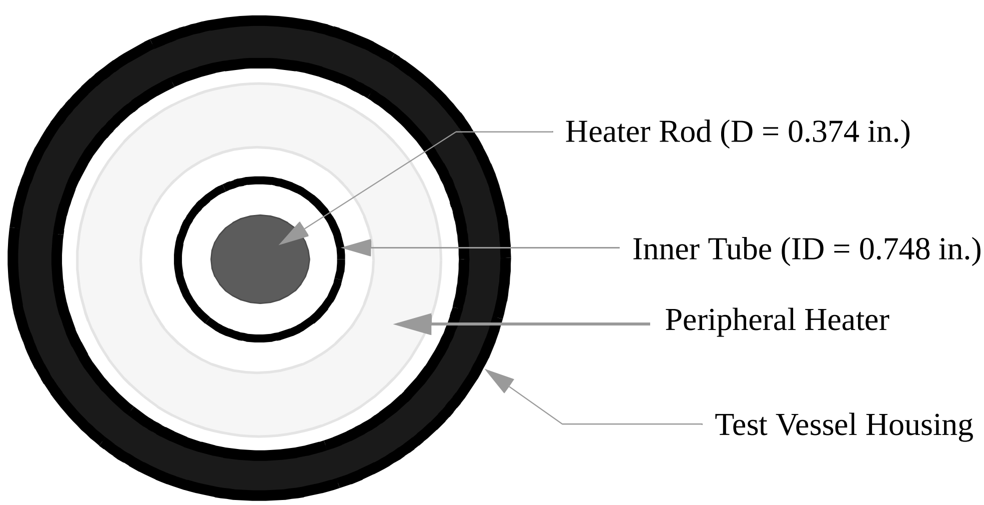

Figure 2. Cross-Section of Test Section

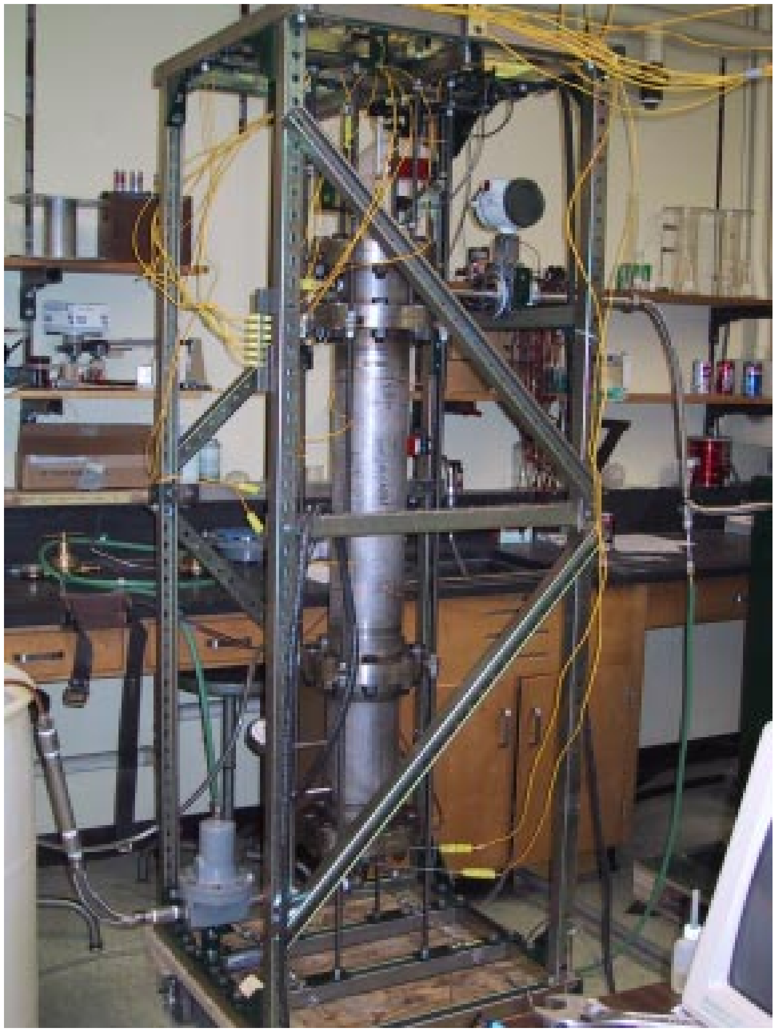

Figure 3. Photograph of Test Vessel.

Results

Results have been obtained at pressures ranging from 40 to 100 psia at a nominal mass flux of G = 50 kg/s-m^2. Inlet quality was varied from 0.8 to nearly 1.0 with inlet subcooling ranged from 40 ^{o}C to saturation.

A review of the initial data showed several problems with the facility. The Inconel inner tube quenched readily at most steam flows and heater rod powers (which controlled the initial temperatures).   The rapid quench of the tube was considered to have strongly influenced the quench rate and quench temperature on the heater rod, as coolant is believed to have preferentially flowed along the tube. Figures 4 and 5 shows transient temperatures on the heater rod and tube. Transient temperatures on the heater rod show a quench front moving down at a rate of 0.01 m/sec. The tube is seen to cool to saturation before quench on the rod occurs. Higher tube temperatures are expected to increase the local quality, slow the downward quench front propagation, and decrease the quench temperature on the heater rod.

In an attempt to control the quench rate on the Inconel tube, two Watlow high temperature ceramic heaters were installed in the region between the inner tube and the vessel housing. Each of these peripheral heaters is 12 inches in length, and have an ID of 1.25 in. This allows the heaters to fit in the gap between the inner Inconel tube and vessel housing without touching the tube or the housing. When power is supplied to the peripheral heaters, the inner tube is radiantly heated and high tube wall temperatures can be maintained. In effect, the peripheral heaters serve as a large "hot patch" to prevent early quench of the tube wall.

Testing with the new peripheral heaters in place showed good control of the tube wall temperature, however after several tests an electrical short circuit developed tripping circuit breakers in AC power supply. The problem was determined to be condensation along the housing inner wall that developed during pressurization of the facility. Electrical leads from the peripheral heaters made contact with the housing wall, and the condensate created an electrical short. This is being resolved, by waterproofing and re-routing the heater leads away from the housing wall, adding insulation to the vessel housing, and revising the experimental procedure. Rather than heating the rig with an initial steam environment, the rig will be pre-pressurized with nitrogen. Steam will not be introduced until the inner wall of the vessel is above saturation. Nitrogen will be valved in and out at the start and completion of tests to suppress condensation.

While pre-mature quench along the Inconel tube may influence results, some important trends were evident in the initial sets of data obtained. Figure 7 shows the effect of inlet quality on quench temperature for a constant mass flux of G = 50 kg/s-m^2. A regression of these data shows a decrease in quench temperature as quality increases. The high scatter is attributed to quench and preferential flow along the tube wall. The values measured were found to be in reasonable agreement with Tmin predicted by the Cheng correlation,

$$T_{min} = 169.66 + 105P + 0.1444G + 3.0347\Delta T_{sub}$$

where T_{min} is in ^{o}C, P in MPa, G in kg/s-m^2, and subcooling in ^{o}C.

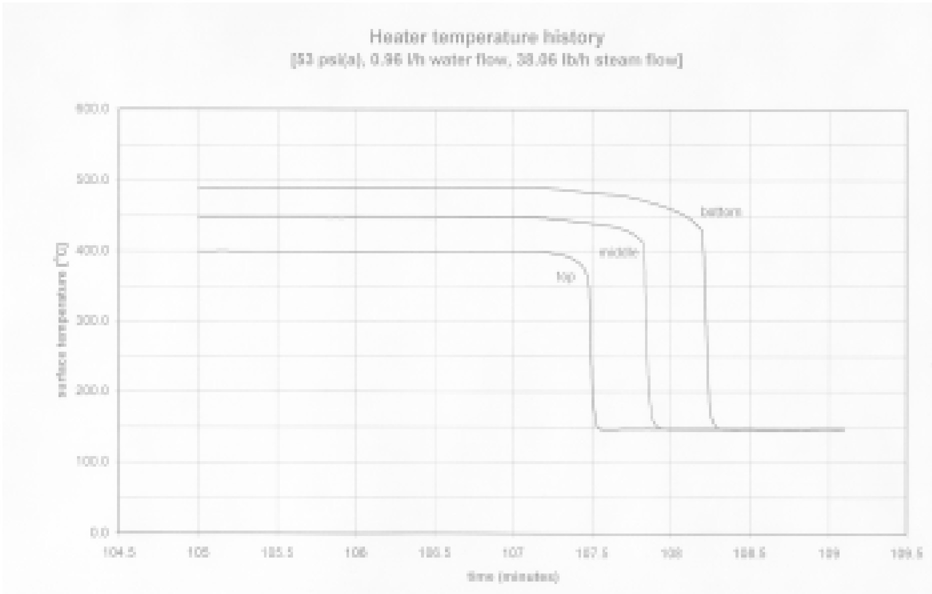

Figure 4. Heater Rod Transient Temperatures

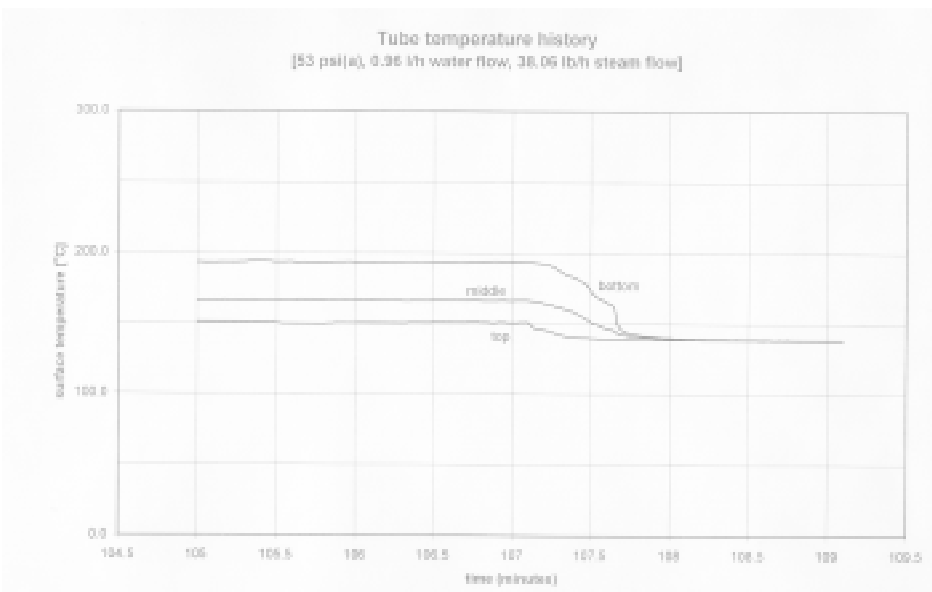

Figure 5. Tube Wall Transient Temperatures

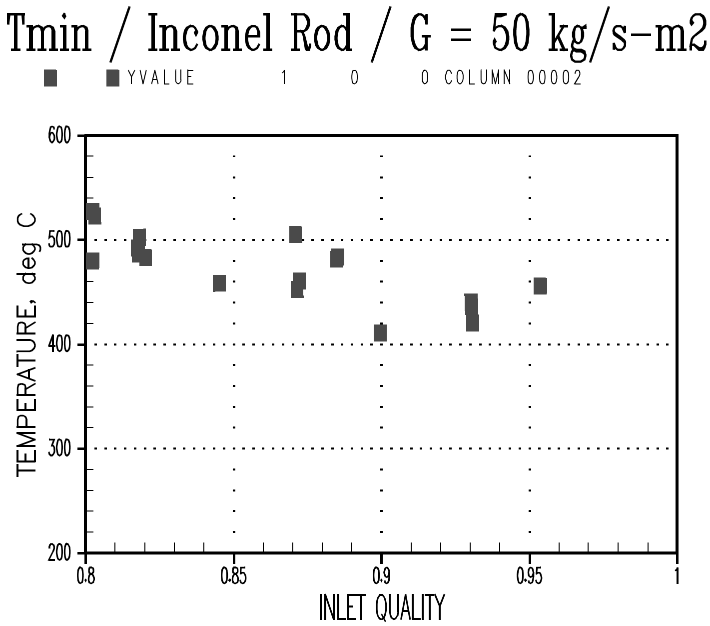

Figure 7. Effect of Inlet Quality on Quench Temperature.

Experimental Facility Revisions

Several modifications are currently being made to the experimental facility before resuming testing. Problems identified in the initial sets of data were: (a) rapid quench of the inner tube, and (b) excessive condensation on the inside of the pressure vessel housing. In addition, the rotameter used to measure the coolant flow rate was difficult to read accurately. To resolve these problems, the following changes will or have been made:

(1) Peripheral heaters are added to maintain a high tube temperature, with electrical leads to be waterproofed and re-positioned away from the housing wall. (Note: Completion expected Jan. 12, 2001.)

(2) A tee will be added to the steam supply line to allow pre-pressurization with nitrogen to suppress condensation on housing wall during initial heatup and to reduce oxidation when facility is not in use. (Note: Completion expected Jan. 12, 2001.)

(3) The coolant rotameter will be replaced with a with turbine flowmeter. (Note: Flowmeter has been received, to be installed by Jan. 19, 2001.)

As of mid-December 2000, all data had been obtained using an Inconel clad fuel rod simulator. Since zircalloy oxidizes readily, it was decided to delay the Zr clad rod tests until the experimental procedures are revised and test facility is modified.

Pool Boiling Quench Tests

A series of tests were conducted to supplement the data being obtained in the blowdown cooling rig. These supplemental tests were conducted in a small, high pressure test vessel that was available in the KSU Thermal-Hydraulics Laboratory and could be applied to the present investigation with relatively minor changes. The high pressure quench test vessel, is capable of operating at pressures up to 3000 psia. An interior high temperature heater is used to heat a small specimen to temperatures exceeding T_{min}. The sample can be quickly plunged into a bath of fluid that is maintained at a specified temperature. These tests and their data are very useful, as they provide information on T_{min} under no-flow, low void fraction conditions and can be used to set up the inverse heat conduction models that will be needed to determine T_{min} from the blowdown quench tests.

The high pressure quench rig was used to obtain quench and minimum film boiling temperatures for 0.374 inch OD (solid) samples of stainless steel and stainless steel sheathed with Zr-4 clad. The Zr clad was obtained from Westinghouse Commercial Nuclear Fuel Division. Figure 8 shows transient temperatures for five independent tests for a sample with a Zr-4 sheath run at a nominal pressure of 80 psia. The quench temperature is the temperature where the “knee” occurs. For this pressure, the quench temperature was approximately 630 ^{o}C. The minimum film boiling temperature is determined from these transient temperatures using an inverse heat conduction calculation. The transient temperatures are used as input to IHCP, an inverse heat conduction package which then calculates surface temperatures and heat transfer coefficients from the embedded thermocouples. Figure 9 shows the surface heat flux versus wall superheat calculated by IHCP for a series of tests using a stainless steel sample (without a Zr-4 sheath). The minimum film boiling temperature is determined from these results, and is indicated by the ΔT at the local minimum heat flux between film and transition boiling.

Tests with zircalloy clad were conducted from 1.01 to 10 bar. Bulk water temperature was maintained at saturation in these tests. Figure 10 shows the variation of quench temperature versus pressure from this series of experiments. The quench temperature (and thus the minimum film boiling temperature) is seen to increase with pressure, and appear to slowly approach a constant maximum value.

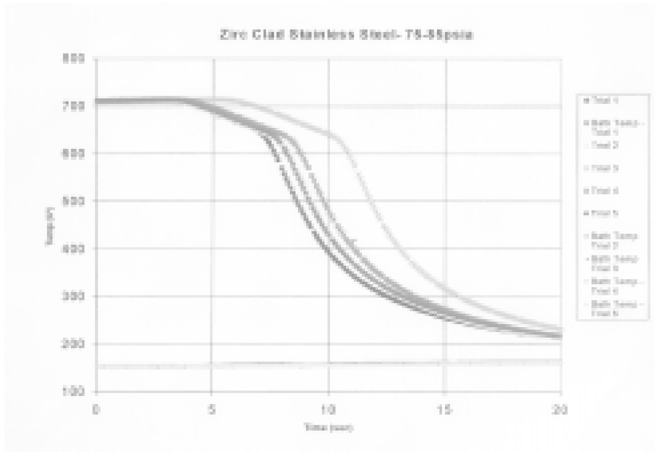

Figure 8.   Pool Boiling Quench Test Transient Temperatures - Zr-4 at 80 psia.

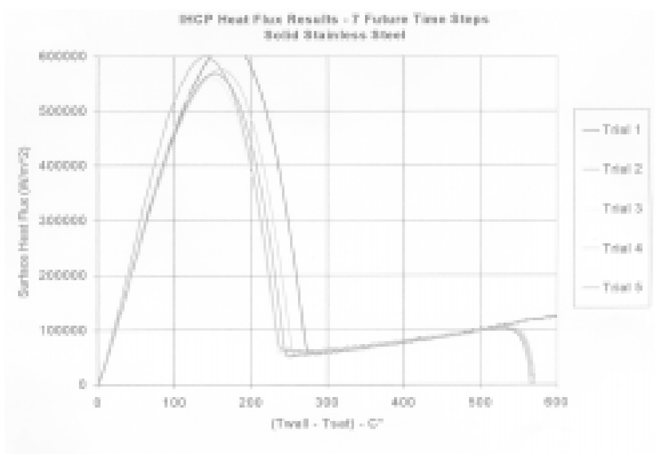

Figure 9.   Wall Heat Flux Determined from IHCP Calculations.

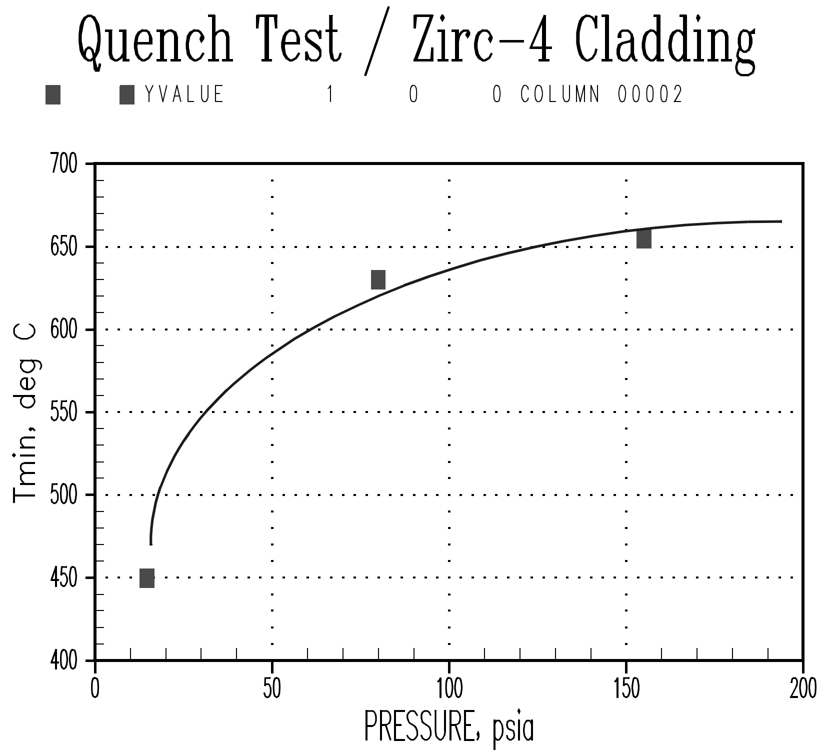

Figure 10. Variation of Quench Temperature with Pressure for Zr-4 Clad.

Neutron Radiography

The purpose of the neutron radiographic work is to develop a method for visualization of a high temperature, high pressure dispersed droplet flow. Model development for film and transition boiling in dispersed droplet flows has been hampered by a lack of information on the flow structure and local void fraction. Initial work in the present investigation focused on "materials testing" and showed that titanium was sufficiently transparent to thermal neutrons so that it could be used in construction of a test section for radiographic inspection of a dispersed droplet flow.

During the second phase of work on this project, a new experimental procedure (required as a facility safety requirement) was written and approved by the University Reactor Safety Committee. One condition of the procedure, and a requirement for Committee approval for open beam port pulsing, was the addition of shielding to insure that public dose rates at the facility boundary are not exceeded. Additional shielding was obtained, and two concrete shield walls were constructed. The shield walls were constructed from high density borated concrete held in place by welded iron frames. The walls can be moved using an overhead crane.

Most recent work has attempted to use "pulse" neutron radiography to visualize a dispersed droplet flow. In a pulse radiograph, a short burst of high reactor power is generated by a large instantaneous reactivity insertion. A pneumatically driven control rod is ejected from the core to gain the required reactivity insertion. Static radiographic tests were used to estimate a pulse strength of $1.60 would be necessary to obtain adequate contrast in the pulse radiograph.

This was verified by static and pulse radiographs of a water column within a titanium tube are shown in Figures 11 and 12 respectively. The pulse radiograph was shot with a reactivity insertion of $1.40, which is well within the facility Tech Spec limit of $2.00.

Pulse radiographs of water droplets falling within a 19.5 mm ID titanium tube have been obtained. The droplet frequency was approximately 150 drops per minute, and the droplet diameters were 3 mm. Contrast of the droplet stream is poor however, due to the long tail of the pulse. After the pulse, the reactor remains at high power (approximately 200 kW) for several minutes. This tail continues to expose the film, and decreases the contrast.

This problem will be resolved by adding a shutter to the beam port. The shutter will be made of borated aluminum, and will shield the experiment being radiographed by dropping a borated plate in a guillotine fashion in front of the beam port. The borated plate will be dropped simultaneously with the pulse, so that the tail is cut off.

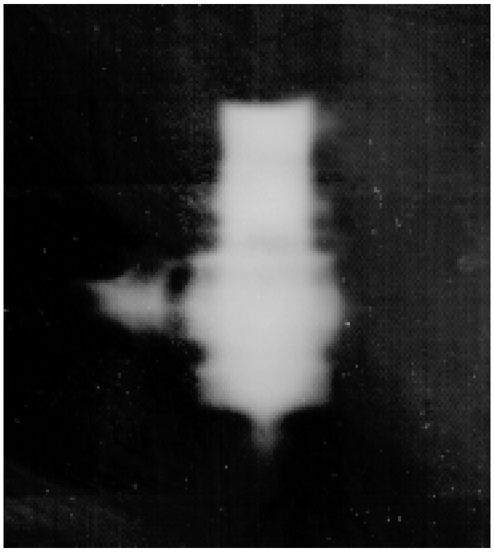

Figure 11. Static Radiograph of Water Column in Titanium Tube. Exposure: 10 min at 10 kW Reactor Power.

Figure 12. Pulse Radiograph of Water Column in Titanium Tube. Reactivity Insertion = $1.40.

SCHEDULE DELAYS

Progress has not been able to keep up with the planned schedule, and work is currently about 2.5 months behind. Recent delays are due to:

(a) A shortage in licensed Senior Reactor Operators (SRO) for the reactor facility. During the Fall Semester, there was only a single SRO. (The NRC imposed a 6 month waiting period before a licensed operator can become an SRO. Thus, the newly hired facility Reactor Manager can not operate and run the reactor in pulse mode until February 2001.) Availability of the single SRO, who is required by our facility license to directly supervise pulse operation, limited the opportunities for performing neutron radiography. Approval of the pulse procedure and construction of the required shielding to insure safe operation also took longer than expected.

(b) Premature quench of the inner tube in the blowdown cooling facility, and the electrical short in the peripheral heaters that were installed to prevent the tube quench delayed testing by approximately 2 months. The delay was due to the time needed for assembly and dis-assembly of the rig, and time required to order and receive the new heaters.

SCHEDULE for PHASE 3

Currently, the facility modifications needed to improve the quality of the data are nearing completion. Testing will resume in January 2001 with the an Inconel clad heater rod. A Zr-4 clad rod will replace the Inconel rod in March and tests will be conducted to obtain Tmin for that material type. Following the modifications currently being made, no additional hardware or facility revisions are anticipated. Testing should proceed rapidly.

Additional time will become available for reactor operation and neutron radiography as two new Senior Reactor Operators (SRO) and three Reactor Operators (RO) and added to our Reactor Facility Staff. (NRC examinations are scheduled for early January 2001, and licenses are expected in early February for the new personnel including the Reactor Manager.) These will augment the current staff (1 SRO and 2 RO), and allow a significant increase in time for experimentation.

The following Table lists planned completion dates for remaining tasks.

Original / Revised Schedule of Tasks

No.	Task	Planned or Actual Start	Revised Finish	Status
1A	Facility Construction / Calibration	03/15/00	06/30/00	Completed
1B	Facility Modifications	11/01/00	01/18/01	In progress.
2	Experimental Testing - Part I (Run heated tests for T_{min}, heat transfer coefficients on Zr and Inconel clad heater rods.)	07/15/00	03/31/01	In progress. Testing suspended pending completion of facility modification.
3	Radiographic Housing Material Test (Determine exposure times, develop reactor pulse technique for various structural materials.)	09/01/99	03/31/00	Completed.
4	Radiograph/Photograph Benchmarking (Run unheated tests in shroud to develop droplet visualization & evaluation methods.)	05/01/00	02/15/01	In progress.
5	Analytical Model Development (Develop “microscale” model for Tmin.)	07/15/99	06/30/00	Completed.
6	Analytical Model Computations/Refinement (Compare to new data, improve models.)	06/15/00	06/30/01	In progress, waiting for data.
7	Experimental Testing - Part II (Heated tests with radiography.)	04/01/01	06/30/01	
8	Documentation and Final Report	06/01/01	08/15/01	